Dennis R. Madison
Vice President - Hatch

Southern Nuclear
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SOUTHERN COMPANY

August 10, 2009

Docket No.: 50-366 NL-09-1221

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
Licensee Event Report
Turbine Trip On High Reactor Water Level
Due To Failed Circuit Board Results in Reactor Scram

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv)(A), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning a reactor scram resulting from a turbine trip due to high reactor water level.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

D. R. Madison
Vice President – Hatch

DRM/MJK/

Enclosure: LER 2-2009-004

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Ms. P. M. Marino, Vice President – Engineering
RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Hatch
Mr. J. A. Hickey, Senior Resident Inspector – Hatch

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Edwin I. Hatch Nuclear Plant Unit 2	05000 366	1 OF 4

4. TITLE
Turbine Trip On High Reactor Water Level Due To Failed Circuit Board Results in Reactor Scram

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
06	23	2009	2009	- 004 -	0	08	10	2009	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 061

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Edwin I. Hatch / Steve Tipps, Principal Licensing Engineer	912-537-5880

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SJ	LC	Y006	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On June 23, 2009 at 03:51 EDT, Unit 2 was in mode 1 with an approximate reactor power of 1710 CMWTh. At this time an automatic reactor scram occurred as a result of a turbine trip due to high reactor water level. Prior to the event the reactor was increasing in power during startup from a recent outage. An instrument which controls reactor water level failed resulting in an increase in reactor water level. The water level increased to a point where the turbine control valve fast closure trip signal was initiated due to high reactor water level. All control rods fully inserted and Reactor Feed pumps tripped. Reactor water level initially decreased to approximately negative 25 inches, due to void collapse. The Primary Containment isolation Valve Group 2 isolation setpoint was reached, and the Group 2 valves isolated. Both the ‘A’ and ‘B’ Reactor Feed Pumps initially tripped, and the ‘A’ feed pump was restarted and used to restore and maintain reactor water level. Reactor pressure reached an upper value of approximately 964 psig. No Safety Relief Valves opened, nor were they required to open, based on the maximum pressure reached. Reactor water level increased to a maximum value of approximately 60 inches above instrument zero, but was restored to normal range.

The cause of this event was the failure of an internal power supply electrolytic capacitor which caused a failure of the DC power supply for the Yokogawa level controller 2C32-K648.

The failed power supply card containing the capacitor was replaced following the event and repetitive tasks have been created to replace this and similar power supply cards at a prescribed interval.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On June 23, 2009 at 03:51 EDT, Unit 2 was in mode 1 with an approximate reactor power of 1710 CMWTh. At this time an automatic reactor scram from a turbine (EIIS Code TA) trip due to high reactor water level occurred. Prior to the event the reactor was increasing in power during startup from a recent outage. At approximately 03:40 EDT, the 2C32-K648 median controller (EIIS Code SJ) failed and no longer tracked actual reactor water level. This controller provides input for the indicated level on the plasma display (EIIS Code SJ) and the 2C32-R600 feedwater master controller (EIIS Code SJ). Reactor water level began to decrease approaching 30 inches as shown by Safety Parameter Display System (SPDS, EIIS Code IU) data however the plasma panel did not indicate this change in level. Reactor water level then began to increase. The High Pressure Coolant Injection (HPCI, EIIS Code BJ) and Reactor Core Isolation Cooling (RCIC, EIIS Code BN) High Reactor Water Level Trip alarms were received. Approximately 30 seconds after receiving the alarm the turbine tripped on the turbine control valve (EIIS Code TA) fast closure trip signal due to high reactor water level. All control rods fully inserted (EIIS Code JD) and Reactor Feed pumps (RFP, EIIS Code SJ) tripped. Reactor water level initially decreased to approximately negative 25 inches, due to void collapse. Primary Containment isolation Valve Group 2 (EIIS Code JM) isolation setpoint was reached and the Group 2 valves isolated. Both the ‘A’ and ‘B’ Reactor Feed Pumps initially tripped, and the ‘A’ feed pump was restarted and used to restore and maintain reactor water level. Reactor pressure reached an upper value of approximately 964 psig. No Safety Relief Valves (SRVs, EIIS Code SB) opened, nor were they required to open, based on the maximum pressure reached. Reactor water level increased to a maximum value of approximately 60 inches above instrument zero but was restored to normal range.

CAUSE OF EVENT

The cause of this event was the failure of an internal power supply electrolytic capacitor on the power supply board which caused a failure of the DC power supply for the Yokogawa level controller 2C32-K648.

During recovery efforts it was determined that the 2C32-K648 controller was not responding to reactor water level increases and was displaying the error code P.error. Per the Yokogawa vendor manual, this is indicative of an internal power supply failure. The P.error code was intermittently displayed during the recovery process.

The controller was removed from service and transported to the Maintenance lab for analysis. Power was applied to the controller, and the P.error code was again displayed. This error was intermittent during the analysis period. Internal inspection of the power supply identified a failed electrolytic capacitor.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Edwin I. Hatch Nuclear Plant Unit 2	05000366	2009	- 004 -	0	3 OF 4

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73(a)(2)(iv)(A) because unplanned actuations of a safety feature system listed in 10 CFR 50.73 occurred. In this instance, a reactor protection system (RPS) actuation resulted in a reactor scram. The reactor protection system (EIIS Code JC) actuated on turbine control valve fast closure when a high reactor water level was sensed.

Fast closure of the turbine control valves is initiated when a reactor high water level condition exists. The turbine control valves close as rapidly as possible to prevent water intrusion into the main steam lines. Valve closing causes a sudden reduction in steam flow that, in turn, results in a reactor vessel pressure increase. If the pressure increases to the pressure relief setpoints, some or all of the SRVs will briefly discharge steam to the suppression pool (EIIS Code BL). In this event the reactor pressure did not reach corresponding setpoint of the SRVs.

Reactor scram and recirculation pump (EIIS Code AD) trip initiation by turbine control valve fast closure prevents the core from exceeding thermal hydraulic safety limits following a main generator or main turbine trip. A reactor scram is initiated on turbine control valve fast closure. The scram, along with the reactor recirculation pump trip system, ensures that the minimum critical power ratio safety limit is not exceeded.

The recirculation pump trip system, upon sensing a turbine control valve fast closure, trips the reactor recirculation pumps resulting in a decrease in core flow. The rapid core flow reduction increases void content and reduces reactivity in conjunction with the reactor scram to reduce the severity of the transients caused by the turbine trip.

In this event, the main turbine tripped as designed in response to the sensed power load unbalance. The turbine trip actuated the reactor protection system and scrammed the reactor. Vessel water level was maintained well above the top of the active fuel throughout the transient. The water level decrease was terminated prior to reaching the automatic initiation set point for HPCI and RCIC. Operations restarted the 'A' feed pump immediately following the reactor scram and subsequent feed pump trip. This pump was used to restore and maintain reactor water level. Therefore no safety system actuations on low water level, including emergency core cooling systems, were automatically initiated nor were any required.

Based on this analysis, it is concluded that this event had no adverse impact on nuclear safety.

CORRECTIVE ACTIONS

The 2C32-K648 controller was replaced.

Repetitive Tasks have been created to replace the power supply for this and similar controllers at a prescribed interval.

A new subsection was added to maintenance procedures 57CP-CAL-226-1 and 57CP-CAL-226-2 to provide instructions for replacement of the Yokogawa power supplies

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Edwin I. Hatch Nuclear Plant Unit 2	05000366	2009	- 004 -	0	4 OF 4

ADDITIONAL INFORMATION

Other Systems Affected: None

Failed Components Information:
Master Parts List Number: 2C32-K648
Manufacturer: Yokogawa (Y006)
Model Number: SCMS-100*E/NPR/MTS/HTB
Type: Level Controller

EIIS System Code: SJ
Reportable to EPIX: Yes
Root Cause Code: B
EIIS Component Code: LC

Commitment Information:
This report does not create any new permanent licensing commitments.

Previous Similar Events:
There are no similar events within the past two years in which a power supply card failure occurred on a reactor water level control instrument.